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Preliminary Analysis of INSC Standard Problem V4: Heat-transfer Under Partly Uncovered Core Conditions at RRC-KI KS-1 Facility

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Introduction

Validation of computer codes for use in safety analysis of VVER reactors has been identified as a topic for a joint research project between the U.S. and the Russian International Nuclear Safety Centers (USINSC/RINSC). The goal of the joint project is to initiate the collaboration between the U.S. and Russian International Nuclear Safety Centers on the validation of computer codes related to nuclear power plant operational and safety analysis. The computer code covered in this project is RELAP5/MOD3.2 [1], which is the code in use for the in-depth safety assessment projects under the US DOE International Nuclear Safety Program.

Earlier phases of the Joint Project included the development of a prioritization report, documenting the safety significant processes and phenomena occurring in VVER type reactors. The resulting VVER prioritization report [2] was used to develop a validation plan for ranking potential Russian tests relative to these safety significant processes and phenomena. The resulting VVER validation plan [3] identifies a number of potential standard problems that could be defined for assessing the capabilities of RELAP5/MOD3.2 in modeling VVER specific safety issues. The working basis for the code validation project is to establish standard problems based on experimental data and then perform the analysis for the standard problems with RELAP5/MOD3.2. The results of the computer analysis will be compared with the experimental results, and the adequacy of the code to model the specific phenomena will be assessed. The standard problems are defined by the RINSC using experimental data from Russian facilities that simulate VVER behavior. The standard problems will be analyzed by U.S. and Russian analysts, and potentially by other international participants. Following the analysis of each specific standard problem by all teams involved, the results will be compared to assess the modeling capabilities of RELAP5 relative to the specific safety parameters thought to be addressed by the standard problem.

The purpose of this paper is to document the USINSC preliminary analysis for VVER standard problem #4 that has been defined in [4]. This standard problem represents heat-transfer under partly uncovered core conditions.

The VVER validation plan [3] identified one phenomenon that was well covered by the data in Standard Problem #4. This phenomenon was **"heat transfer in a partially uncovered core"**. The adequacy of RELAP5/MOD3.2 to model the above phenomena is limited by the measurements available in the KS-1 facility and the facility design. As identified in [4], the test section is such that heat losses to the ambient air (due to condensation dominated heat-transfer) exceed the heat injected into the coolant through the rod simulators. This leads to unsteady or quasi-steady behavior of the system resulting in slow pressure decreases and significant changes in the core liquid coolant level. Because the RELAP5 initialization does not simulate the actual sequence of events preceding the data collection window, the correct modeling of the unsteady conditions during the experiment is difficult and must be verified carefully. The test results do not include coolant flow measurements, which adds to the difficulty in initializing the RELAP5 model.

The following sections of this paper include a description of the KS-1 facility, a description of the test being studied, a description of the RELAP5/MOD3.2 input model developed by ANL, summary of preliminary analysis results, followed by conclusions.

Description of KS-1 Test Facility

VVER Standard Problem #4 is based on heat-transfer under partly uncovered core conditions in the KS-1-VVER test section of the KS test facility at RRC Kurchatov Institute (KI). Experimental section KS-1 was developed within the KS test facility for the purpose of modeling thermal-hydraulic processes in a VVER core under Small Break LOCA conditions. Experimental section KS-1-VVER-1000 has been designed for modeling of heat transfer processes in a partly uncovered VVER-1000 core at small residual heat power under conditions of natural coolant circulation stagnation and for medium and low pressure ranges. The test facility was designed to investigate both separate effects and integral thermal and hydraulic processes in sections of the primary circuit.

Main parameters of the experimental section

The KS-1-VVER-1000 test section is a semi-integral, one loop model of a VVER primary system. A schematic of the test section is shown in Fig.1. The experimental section includes models of all main elements of a VVER type reactor, including the upper plenum (UP), the lower plenum (LP), the hot and cold legs of the primary coolant loop and the steam generator (SG). The VVER horizontal tube bundle SG is simulated by two un-insulated pipe sections with passive heat removal. The SG simulator only qualitatively simulates the hydrodynamics and heat transfer processes under steam condensation in the horizontal, steam generator tube bundle. For this reason the experimental section KS-1 is a semi-integral model of a VVER primary system. The core simulator, depicted in Fig. 2, incorporates a fuel assembly (FA) model consisting of 19 electrically heated tubes with a diameter of 9 mm and a heated zone height of 2.505 m.

Test Description

In 1991, investigations of unsteady boil off, core uncovering and heat-transfer in partly uncovered FA models were conducted under quasi-steady and unsteady conditions at experimental section KS-1. Fuel pin transient temperatures were measured for a VVER-1000 FA model at small values of residual heat power under conditions of steam-condensate circulation for different levels of two-phase mixture in the core for quasi-steady flow regimes. The results of six experiments were chosen to formulate standard problem V4 to evaluate the adequacy of RELAP5 for modeling the conditions corresponding to a partly uncovered core at residual power. The tests selected (KS-1-18-1, KS-1-18-2, KS-1-19-1, KS-1-19-2, KS-1-33-1, and KS-1-34-1) represent variations in mixture level, heat flux, pressure, and steam flow rates.

The goals of the experiments included:

1). obtaining test data on axial and radial temperature distributions within the fuel rod simulators under various power and coolant levels in the core model,

2). determination of the influence of thermal and hydraulic processes in the primary loop component models under partly uncovered core conditions,

3). determination of the influence of loop pressure on the temperature distribution of the rod simulators under constant values of power and core mixture level for partly uncovered core conditions.

Experimental Methodology

Experiments on modeling of quasi-steady regimes of steam-condensate circulation and partly uncovering of the core were conducted under various pressures, coolant temperatures in the lower plenum model, different mixture levels in the core and various power levels. To establish the appropriate experimental conditions, preliminary heating of the coolant water and pipeline metal to the required temperature at the core model inlet was performed under forced coolant circulation. The VVER loop model was next isolated from the forced circulation circuit of the KS facility. A smooth increase of the FA power was imposed to establish a single-phase (non-boiling) natural circulation regime. Heating of the coolant water increased the coolant pressure to 80-90 bar. Coolant drainage from the SG tube bundle simulator to the expansion tank was initiated through valve 1. After the onset of boiling at the core outlet, a natural circulation, boiling regime was established in the loop under slightly greater pressure and coolant inlet temperature than specified for the test. Some steam was released from the tube bundle simulator to the expansion tank through valve 1.

Full draining of the UP model was made after the pipeline metal temperature in the upper part of the VVER circuit model reached saturation conditions near the specified values of pressure and coolant temperature at the core inlet for the individual tests. Also the core mixture level was established by draining water from the lower pipeline to the expansion tank through the valve 2.

After establishment of the appropriate inlet coolant temperature and mixture level in the core model, the main test was conducted for investigating heat-transfer in the partly uncovered FA model under conditions of steam-condensate circulation with constant FA power under a quasi-steady regime. During the tests, valves 1, 2, and 3 were closed. The quasi-steady behavior of these tests is due to the condensation dominance of the test loop design. The condensation dominance is the result of heat losses from the test loop exceeding the electrical power added to the core simulator rods, thereby causing a slow decrease in system pressure, core mixture level variations and variations in rod simulator wall temperatures.

Test data was recorded for the following parameters: the loop pressure at the core model outlet, differential pressures in the core region and around the primary loop, coolant temperatures at the core inlet, outlet and downcomer, 33 rod simulator wall temperatures at combinations of 14 radial and 20 axial locations, as well as current and voltage drop across the heated channel (representing the electrical power of the FA model). Coolant flow rates were not measured due to insufficient measurement ranges of the instrumentation.

RELAP5 Model and Experiment Simulation

A preliminary RELAP5 model for simulating the V4 tests was developed incorporating models for the core, upper plenum, loop hot leg, steam generator, loop cold leg, downcomer, lower pipeline, and lower plenum. The nodalization of the RELAP5 model is depicted in Fig. 3. The nodalization of the core and core annulus regions are depicted in more detail in Fig. 4. The core model consisted of 20 axial volumes with a heat source representing 19 heated rod simulators. The core annular region also consisted of 20 axial volumes which were connected thermally (representing the FA shroud) to the 20 axial core volumes.

The boundary conditions imposed on the RELAP5 simulation model included a heat source in the heated core section and estimated heat-loss coefficients for different parts of the test loop based on results of a separate steady state test. The computations were conducted by running for 200 sec. with no heat source and no heat losses, then increasing both the heat source and the heat losses linearly to their maximum values from 200 – 600 sec. After 600 sec. the heat source and the heat loss coefficients were kept constant. Because the heat losses exceed the heat addition to the system, the loop pressure decreases slowly. The computed core outlet pressure was compared with the experimentally observed values, and the heat-loss coefficients were adjusted so that the computed core outlet pressure decrease rate matches the decrease rate observed in the experiment. The computational results were assumed to reach the initial experiment conditions when the computed core outlet pressure reached the corresponding value measured at the beginning of the experimental time window. The initial computational conditions were selected such that the computed core and downcomer water levels matched the observed values at the beginning of the experimental time window. The coolant flow-rate was driven by natural circulation.

RELAP5 Analysis Results for Test KS-1-19-2

A preliminary analysis of experiment KS1-19-2 was performed, with the goal of determining the influence of initial and boundary conditions on the computed results. These calculations investigated the impact of initial coolant temperature distribution and coolant level in the fuel assembly, the impact of the maximum heat loss multiplier and the role of the power and heat-loss increase rate on the RELAP5 results.

General Observations

Through a judicious selection of initial and boundary conditions it was possible to obtain reasonable agreement with much of the experimental data, but not all. Disagreements between the experimental data and computed results were noticed in the case of upper-core fuel rod temperatures, which were generally higher than the measured values. The behavior of the calculated upper-core fuel rod temperatures during the experimental data collection window is highly dependent on the initial core coolant temperatures, which in turn affect the behavior of the core liquid level. The core liquid level at the beginning of the experiment window was in reasonable agreement with the experimentally observed level (1.78 m). However, the liquid level has not yet reached a quasi steady-state condition during the experiment time window. In most cases examined the liquid level is first increasing for about 200 sec. after the initiation of the power and heat loss increase, then decreases during the experiment time window, and appears to reach a quasi steady-state level after 1000 sec. In all cases examined the experimental time window ends before 1000 sec.

The slope of the calculated core outlet pressure curve is directly influenced by the heat-loss multiplier C, and is not sensitive to the initial core coolant temperature distribution. The range of values C=2-2.15 provided reasonable agreement with the experimentally observed pressure decrease rates.

Base Case Results

Figures 5 and 6 show the calculated inner fuel pin temperatures and the collapsed liquid level in the core, respectively, for the base case (Case 12) which was initiated with the core channels being filled with liquid up to 1.78 m and core coolant temperatures varying between 488 K – 511 K from the core inlet to the liquid-vapor interface. The heat-loss multiplier C was 2, and the core outlet pressure magnitude and slope were in agreement with the experimental values, as shown in Fig. 7. Fig. 8 presents the measured inner fuel rod temperatures in the upper half of the core. Qualitative agreement is observed, with both calculated and measured temperatures at and above 2.135 m considerably higher than the coolant saturation temperature, indicating the presence of lower heat transfer coefficients between cladding and coolant. Both calculated and measured temperatures at lower levels are close to the coolant saturation temperature, indicating the presence of boiling liquid in the channel and corresponding high heat transfer coefficients between cladding and coolant. Figures 9-13 compare the calculated inner rod temperatures in the upper half of the core with the corresponding measured values. The calculated rod temperatures in the upper core region, which contains superheated vapor, are higher than the measured values. Furthermore, these temperatures are still increasing, while the measured temperatures show a decreasing trend. The explanation for this discrepancy is likely related to the core liquid level behavior during the experimental time window.

As shown in Fig. 6, the calculated core liquid level, which has increased initially during the transient, is decreasing during the experiment time window. Only limited information was available about the liquid level behavior during the experiment, indicating that the liquid level was increasing initially at a rate of 2.1 mm/s. An examination of measured temperatures shown in Fig. 8 indicates that this rate of increase was not sustained during the experiment window, because it would have led to rod temperature decrease (quenching) at level 2.135 m and higher by the end of the measurements. An evaluation based on the progression of the quenching front, using the rod measured temperatures, indicates that between 50 – 100 sec the liquid level was increasing at a rate of ~ 1.6 mm/s. This points to a non-linear behavior of the core liquid level during the experiment window. Such behavior is supported by the computed core liquid level shown in Fig. 6, which indicates that the liquid level first increases between 300-500 sec, and then decreases during the 500-1000 sec interval, which includes the experiment window. Fig. 14 shows the calculated core liquid level for the base case up to 2000 s. , suggesting that the liquid level is still in a transient state during the experiment window, and reaches a quasi steady-state after 1000 s.

The transient state of the core liquid level, combined with the fact that the rod temperatures in the voided core region are directly influenced by the liquid level position, indicates that more attention should be directed toward comparing the calculated core liquid level with the level inferred from experimental data. Agreement between the calculated and experimental liquid levels is necessary if one wants to evaluate the ability of RELAP5 to model the heat transfer between rods and the coolant vapor.

Parametric cases.

A number of variations of the base case were run in order to evaluate the impact of various parameters on the upper core rod temperatures and the core liquid level. The upper core rod temperatures during the experiment window were found to be sensitive to the core coolant temperatures used to initiate the calculation. Figures 15 and 16 illustrate the calculated rod temperatures and core liquid level for Case 17, a variation of the Base Case, in which the initial core coolant temperatures were 511 K in all liquid volumes, close to the saturation temperature. The general behavior of the core liquid level remains similar to the corresponding results obtained in the base case, but the behavior of the calculated rod temperatures is different, with the rod temperatures in nodes 16 and 17 indicating that dry-out is occurring during the experiment window. These differences are attributed to changes in the core liquid level behavior, combined with a shift of the experiment window. The effect of time shifts of the experiment window, which should have a small effect in a quasi steady-state situation, are amplified by the transient core liquid level behavior. Other parametric cases using different core coolant initial temperatures show a similar trend.

Summary and Conclusions

The approach used to model the experimental conditions with RELAP5 provides a reasonable description of the phenomena and conditions which existed during the data collection period. However, the preliminary analysis shows that the calculated upper-core rod temperatures are only in qualitative agreement with the data. These temperatures are directly affected by the core liquid level behavior during the experiment window, which in turn is influenced by the core coolant temperatures and level used to initiate the calculation. Because the RELAP5 initialization does not attempt to simulate the sequence of events preceding the data collection window, the core coolant temperatures and level at the start of RELAP5 calculations are not based on experimental data but are selected to match certain parameters at the beginning of the experiment window. Furthermore, the rod temperatures are influenced by a transient liquid level determined by the computational transient preceding the window, which does not necessarily model the experimental sequence of events. Due to these uncertainties, it is not possible to draw definite conclusions about the code performance. Additional information about initial and boundary conditions, if available, and a comparison of the calculated and experimental core liquid level could help improve the agreement between the calculated results and the experimental data, increasing our confidence that the experimental conditions are modeled correctly.

References

1. RELAP5/MOD3 Code Manual, NUREG/CR-5535 (INEL-95/0174), Volumes I-V, Idaho National Engineering Laboratory, Idaho Falls, Idaho (June 1995).
2. “Computer Code Validation for Transient Analysis of VVER and RBMK Reactors*: Validation Prioritization on VVER”*, International Nuclear Safety Center of MINATOM of Russia, deliverable 5 of Phase 1 of Joint project 6, Moscow, Russia (1998).
3. “Computer Code Validation for Transient Analysis of VVER and RBMK Reactors: A *Final RELAP5 Validation Plan for Application to VVER”*, International Nuclear Safety Center of MINATOM of Russia, deliverable 3 of Phase 2 of Joint project 6, Moscow, Russia (1998).
4. “Computer Code Validation for Transient Analysis of VVER and RBMK reactors: *Standard Problem INSCSP-V4 Definition Report”*, International Nuclear Safety Center of MINATOM of Russia, Task 2, Deliverable V4-1 of Phase 3 of Joint project 6, Moscow, Russia (2000).

Figure 1 Schematic of KS-1 VVER Test Loop

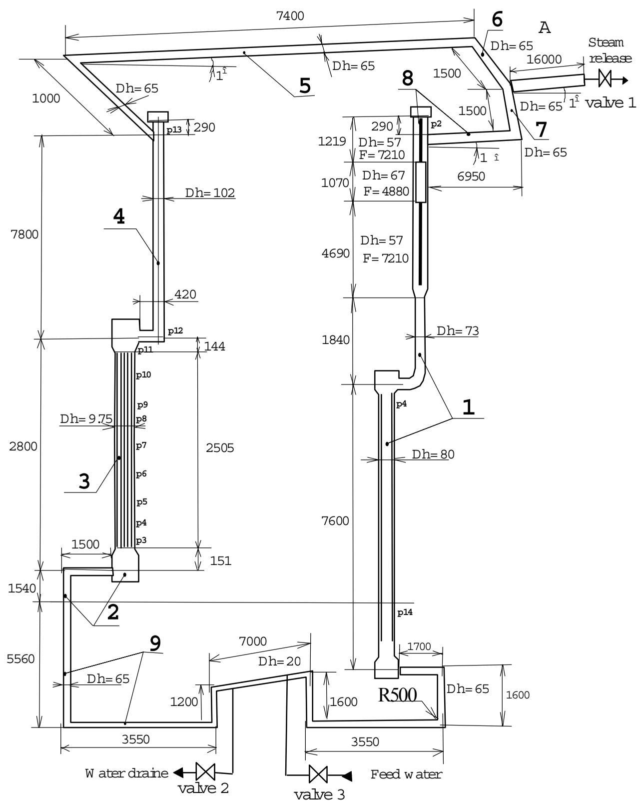

Figure 2 KS-1 VVER Core Simulator

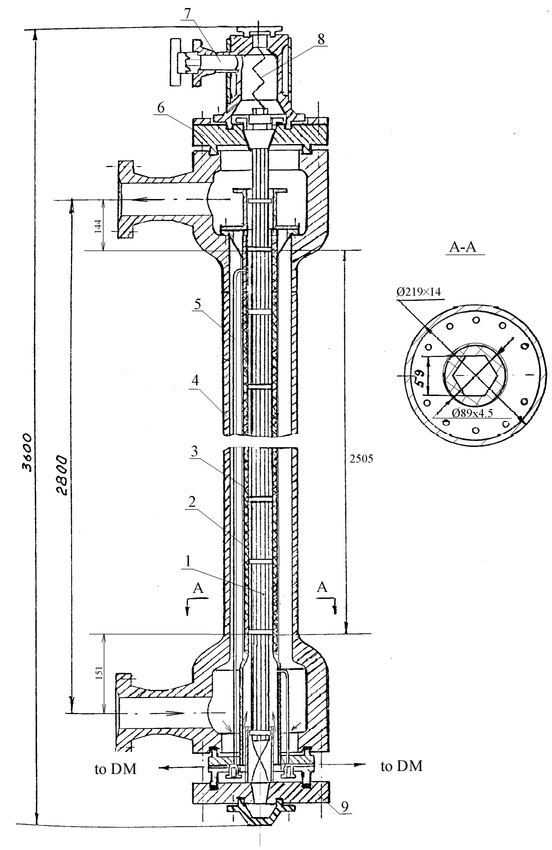

Figure 3 V4 RELAP5 Loop Nodalization

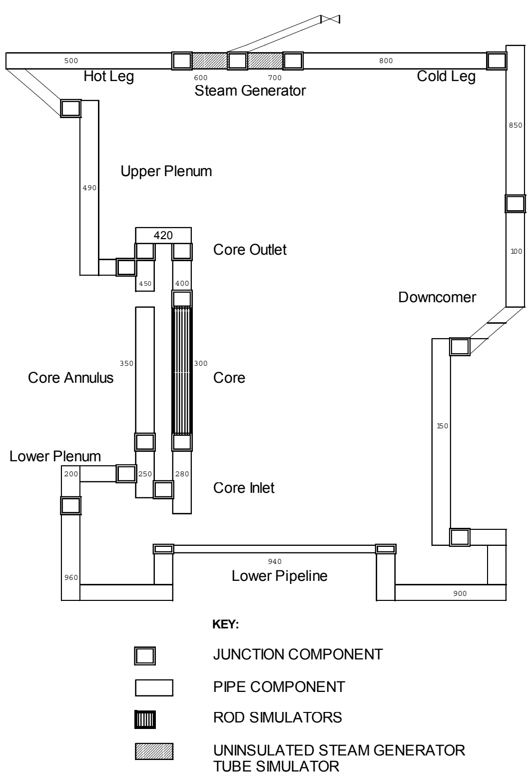

Figure 4 V4 RELAP5 Core Nodalization

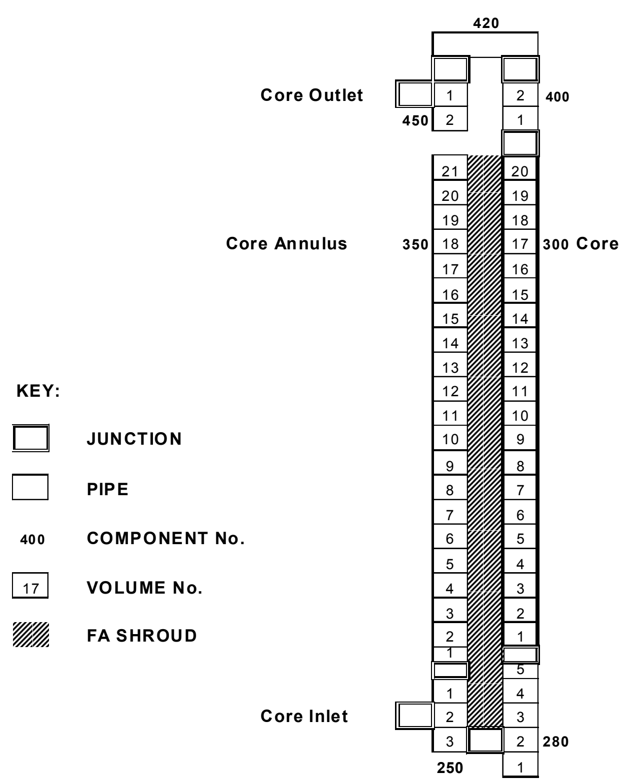

Fig. 5

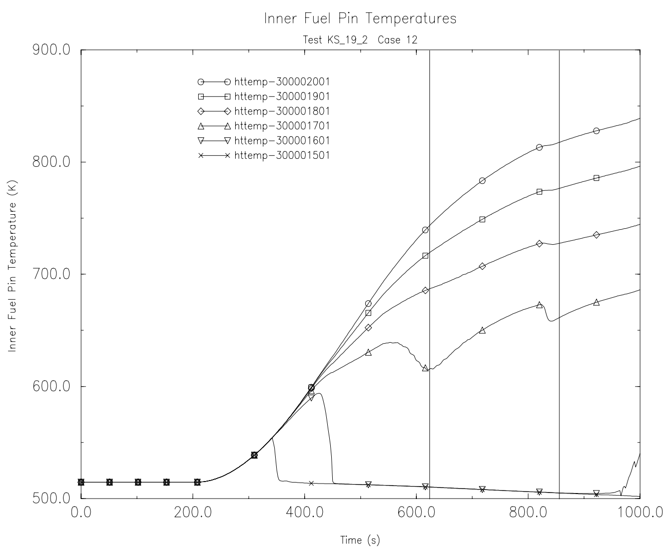

Fig. 6

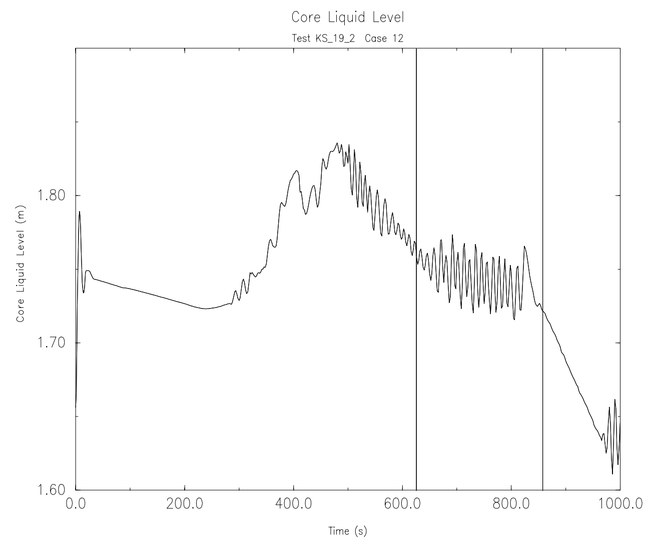

Fig. 7

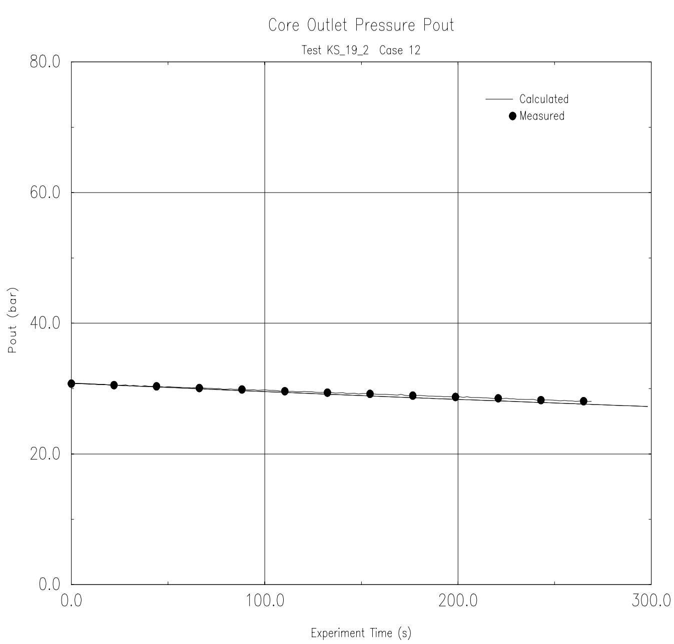

Fig. 8

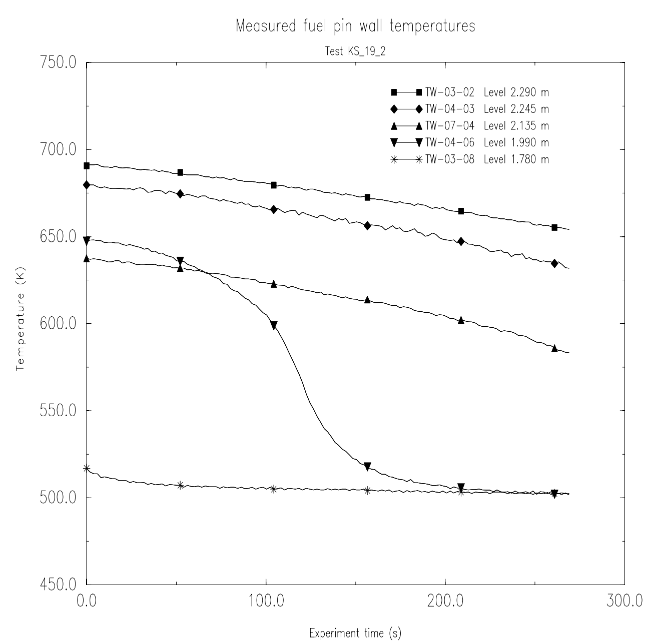

Fig. 9

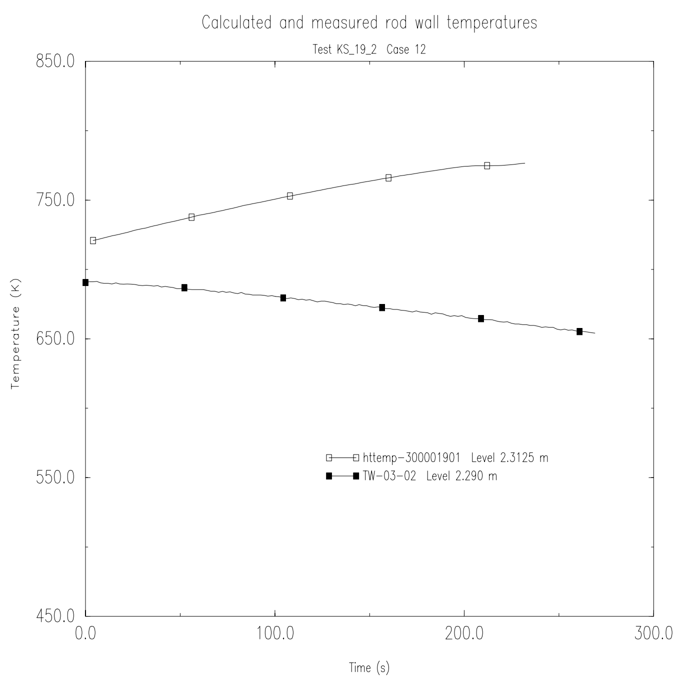

Fig. 10

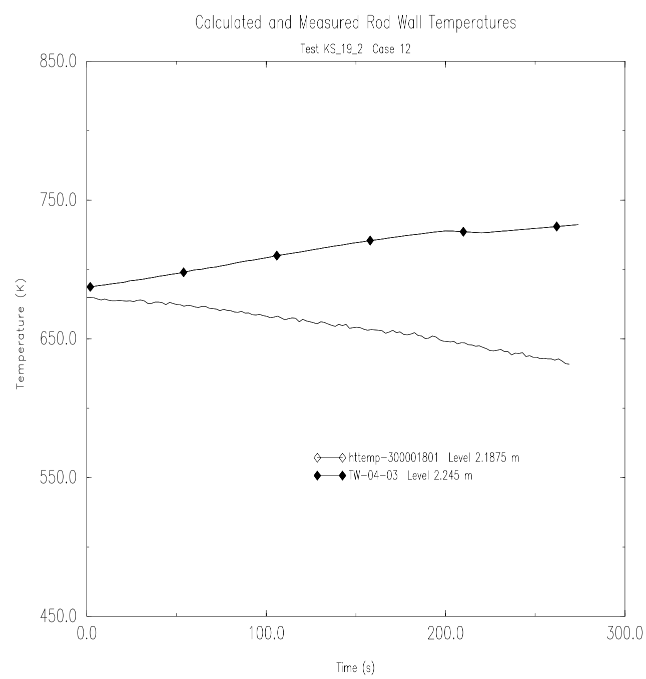

Fig. 11

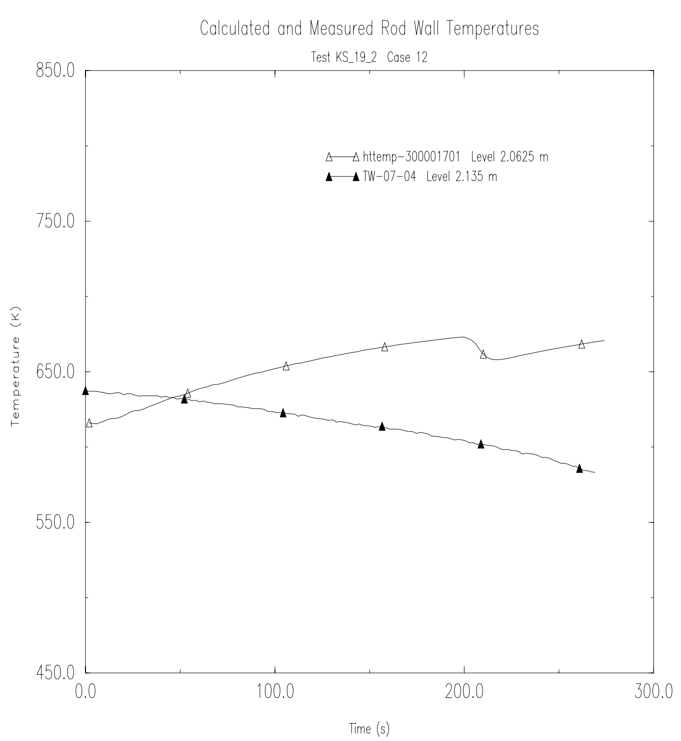

Fig. 12

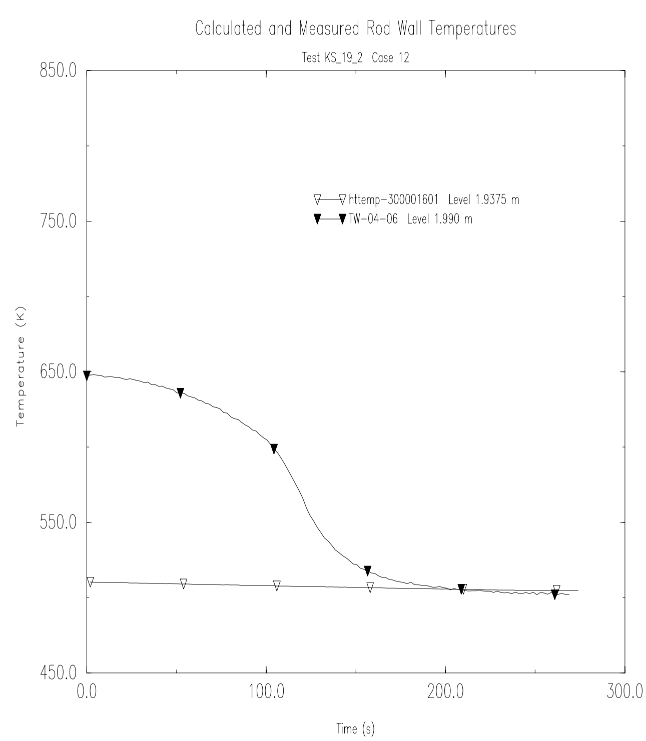

Fig. 13

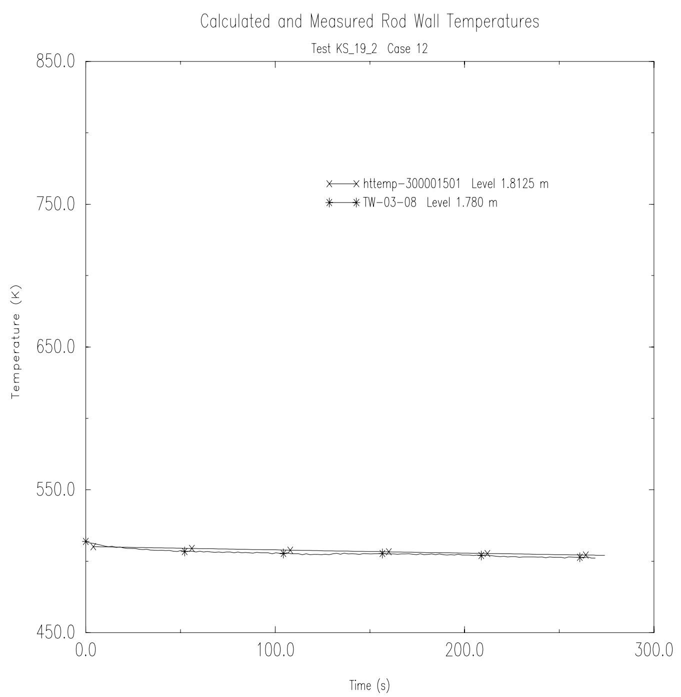

Fig. 14

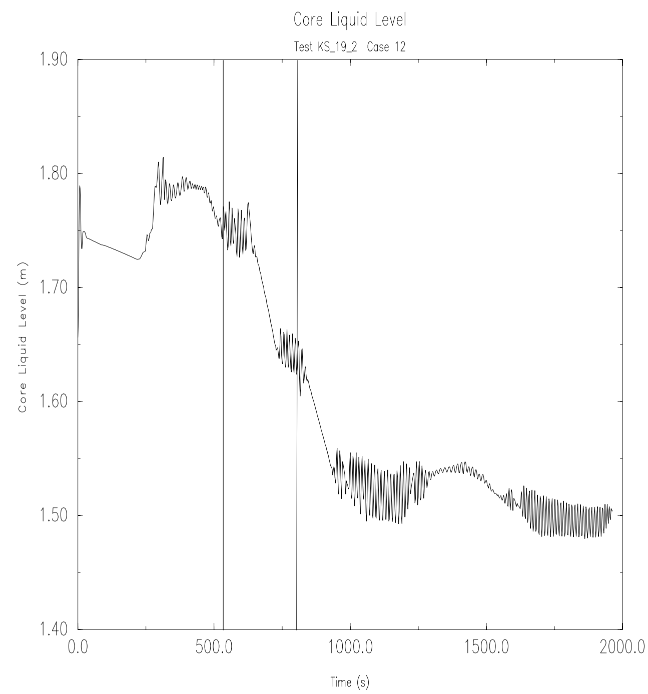

Fig. 15

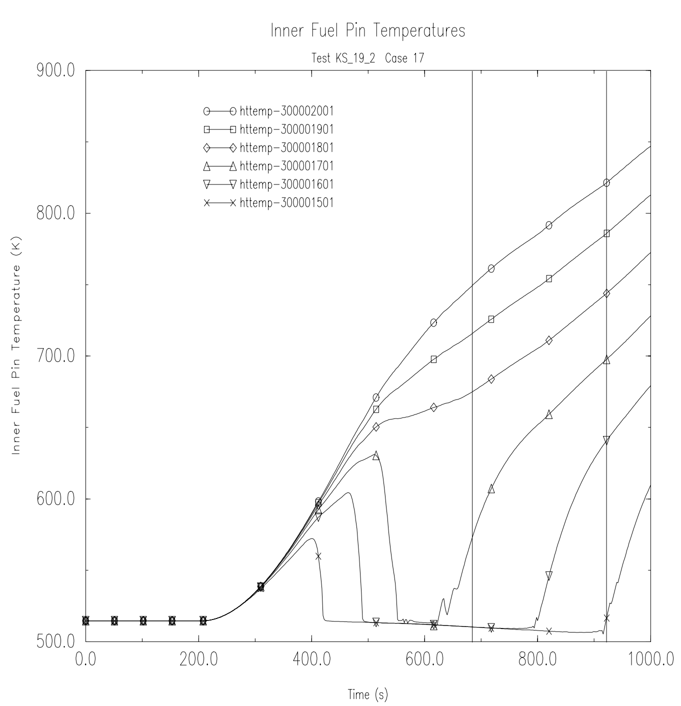

Fig. 16